Overview of Recent JET Results and Future Perspectives

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Abstract. Recent JET results show progress in two directions: consolidation of the ITER reference scenario, the ELMy H-mode, and development of regimes with internal transport barriers (ITBs). The beneficial effect of high triangularity in ELMy H-mode - one of the features of the new ITER design - has been confirmed to extend the operation limits. Indeed three different ELMy H-mode scenarios are shown to allow operation at or above the Greenwald density limit with limited or no confinement degradation: (1) operation at high triangularity (up to δ=0.47) with controlled gas puffing; (2) operation well above the L-H transition threshold (P_{in}>2.5P_{L-H}) with controlled gas puffing; (3) impurity seeding. The level of confidence in the ITER ELMy H-mode reference scenario has therefore increased further and is high; research should now focus towards milder ELMs. Very significant progress in LHCD coupling has been achieved, with routine power in the 2-4 MW range. LHCD preheat leads to efficient control of the q profile, giving access to strongly reversed shear plasmas, with very broad ITBs on T_i, T_e and n_e, achieved at reduced power thresholds. ITBs are being developed, either towards high performance ($H\beta_N$~7.5 achieved) or towards steady state (full non-inductive current drive achieved during 1second).

1. Introduction

Since 1 January 2000 the JET facilities are used in the frame of the European Fusion Development Agreement (EFDA). This multilateral agreement provides a framework under which the JET scientific programme is fully integrated with other EU fusion activities, with a reinforced focus towards ITER. The scientific and technical responsibilities are distributed in the EU laboratories who work in a co-ordinated way to conduct the programme [1, 2].

The ongoing experimental programme gives strong priority to experiments which could have an impact on the ITER design: plasma shaping; operation near boundaries (β_N, n_{GW} limit, H-mode threshold etc…); divertor characterisation and modelling; effect of neoclassical tearing modes, etc... Internal Transport Barrier (ITB) regimes are also studied in view of developing potential steady state-scenarios. The results presented here are a selection of the first results obtained during the experimental campaigns which started under this new framework on 31 May 2000: operation at high power high density ELMy H-mode (section 2); high triangularity ELMy H-mode (section 3); impurity seeding (section 4); neoclassical tearing mode studies; (section 5), improvement in Lower Hybrid coupling (section 6); progress in Internal Transport Barrier regimes (section 7).

The use of the facilities under EFDA will continue during the period 2001-2002 (section 8). On a longer time-scale, approval has been granted to an increase of heating power up to 50 MW (section 9).

2. Operation at High Power, High Density in ELMy H-mode

The demonstration of the simultaneous achievement of high density (f_{GW} > 0.85; $f_{GW} = n/n_{GW}$ where the n_{GW} limit is $n_{GW} = I/\pi a^2$ [10^{20} m^{-3} MA m^{-2}]) and good confinement at relevant values of β_N (β_N ~ 1.8) is of great interest for the predictions of ITER performance in ELMy H-mode. To this effect, a series of experiments were performed in JET with the MkII-Gas Box Divertor at 1.9MA/2.0T (q_{95} ~3.4), with moderate shaping (triangularity δ~0.33). The plasma was heated by neutral beams (P_{in} ~ 15-17 MW) and the density increased gradually from pulse to pulse with Deuterium gas fuelling (up to 4.5 10^{22} s^{-1}). Previous experiments [3] had shown that an increase in fuelling rate results in a degradation of confinement. In the present experiments, this degradation is avoided as shown in Fig. 1: the plasma energy content is almost constant (and close to that of the reference discharge without gas fuelling) up to the n_{GW} limit, with β_N in excess of 2 up to the highest density.

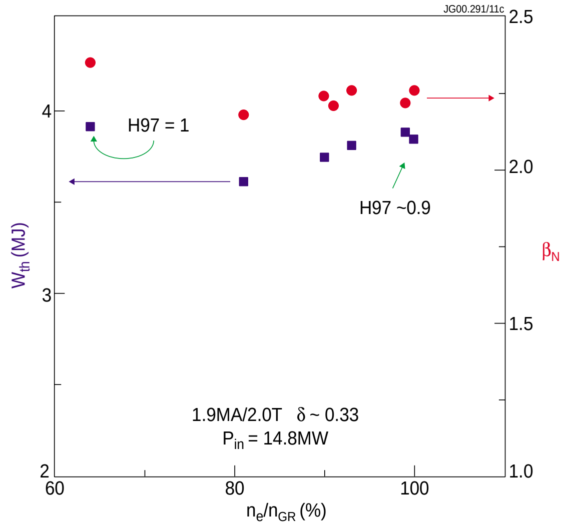

Fig. 1: thermal plasma stored energy W_{th} and β_N (total) as function of the plasma density (normalised to n_{GW}). The H97 confinement enhancement factor is shown for the first point (no gas fuelling, n~ 0.6 n_{GW}) and for the last point of the scan (strong gas fuelling, n=n_{GW}). Each point of the graph represents a separate discharge.

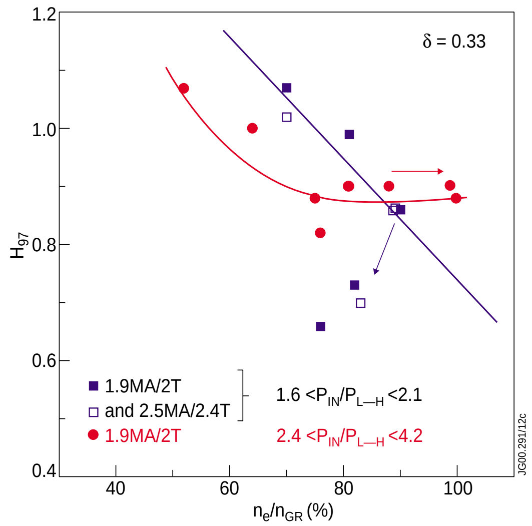

Fig. 2: Energy confinement enhancement factor as a function of the plasma density (normalised to n_{GW}). Each point represents one discharge. The gas scans included are: 1.9 MA/2.0T, high power (14.8 and 17 MW NBI, circles), and lower power (8MW, full squares) and 2.5 MA/2.4 T with 14.8 MW (open squares). The variation in the margin above the threshold for each scan is due to the density.

TAB. I: JET ELMY H-MODE ACHIEVEMENTS COMPARED TO ITER REQUIREMENTS

	JET Pulse No. 50847 2.4 MA/2.4T	JET Pulse No: 52022 2.5 MA/2.7T	ITER-FEAT
Scenario	High P/P_{L-H} (section 2)	High δ (section 3)	
H_{97}	0.93	1.0	1.0
β_N	1.78	1.7	1.81
n_e / n_{GW}	0.89	1.0	0.85
Z_{eff}	1.4	1.7	1.7
P_{rad}/P_{tot}	0.47	0.51	0.58
κ, δ	1.7, 0.33	1.7, 0.43	1.84, 0.5
q_{95}	3.3	3.0	3.0
τ_{pulse} / τ_E	10*	10*	110

*limited by duration of neutral beam heating

The important change with respect to previous data is an increase in the ratio of input power, P_{in}, to the L to H-mode power threshold [4], P_{L-H}, from $P_{in}/P_{L-H} < 2.1$ to $P_{in}/P_{L-H} > 2.4$. This is illustrated in Fig. 2 and Table 1: at high P_{in}/P_{L-H}, good confinement is achieved up to n_{GW}, at low $P_{in}/P_{L\text{-}H}$ the confinement is degraded close to L-mode values in the case of the strongest fuelling. The high density, high confinement discharges maintain Type I ELMs (at high frequency, ~40Hz) and the energy loss per ELM is only 3% of the plasma stored energy, approaching the ITER requirements. Some peaking of the density profiles is commonly observed in steady state ELMy H-modes in JET. For the discharges in Fig. 1, $n_{max}/\langle n\rangle_{vol}$ ~1.3 is maintained up to n_{GW}. The plasma Z_{eff} is very low, ~1.4. The extrapolation to ITER of the achieved plasma parameters indicate that the basic Q = 10 performance of the reference scenario [5] could be exceeded.

3. Influence of Upper and Lower Triangularity Variation on High Density H-mode Operation

In JET ELMy H-mode discharges, high average triangularity has been shown to improve plasma performance, measured by the product $f_{GW} \times H_{ITER89p}$ [1]. Investigations into the triangularity dependence of the pedestal pressure in ASDEX-U supported this result, but could not resolve the individual roles of upper and lower triangularity (δ_{up}, δ_{low}) due to operational constraints [6]. To separate the dependence of ELMy H-mode confinement on δ_{up} and δ_{low}, special plasma shapes were developed at JET, including ITER relevant high triangularity discharges. Fig. 3a shows the δ_{low} and δ_{up} variation. These discharges were obtained with no or moderate gas puff in the Mark IIGB divertor. Table 1 shows the characteristics of one of the best pulses.

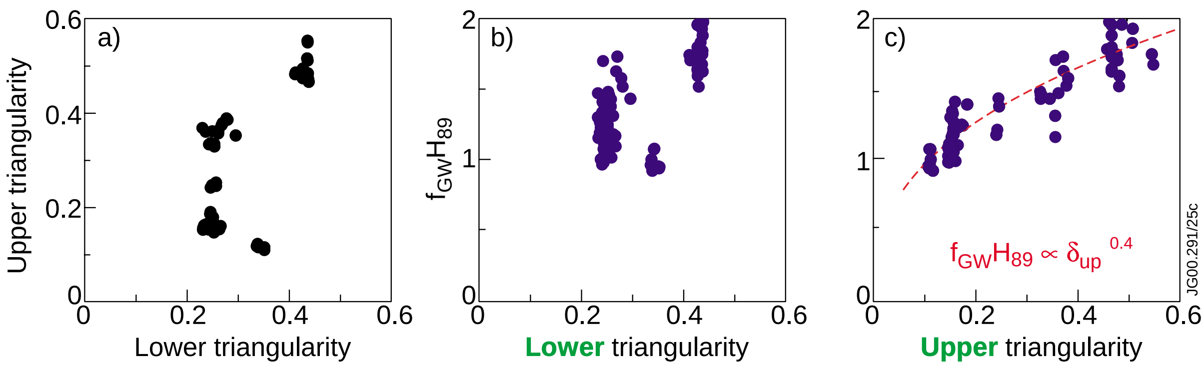

Fig 3: Data from shape scan: type-I ELMy H-mode discharges, with I_p ~2.5 MA, q_{95}~3 and P_{NBI}~13 MW: (a) Variation of upper and lower triangularity; (b) $f_{GW}{\times}H_{ITER89p}$ versus lower triangularity, time-averaged over several ELMs; (c) Same versus upper triangularity.

The data shown in Fig. 3 represent a sub-set of the data base with only small variations (±15%) in plasma current, field and heating power to focus on the effect of the triangularity. These dedicated discharges point to the upper triangularity as the leading shape parameter (see Figs. 3b and c), with a fit for the performance parameter $f_{GW} \times H_{ITER89p} \sim \delta_{up}^{0.4}$, while δ_{low} seems to have a more limited effect. The direct effect of the triangularity is most pronounced on f_{GW}, providing an increase in density with δ_{up}, while the confinement does not degrade as seen in H-mode density ramps using gas puffing. The reason for the different behaviour of the δ_{low} and δ_{up} remains to be clarified. The stabilising effect of the magnetic topology close to the X-point might be invoked, but some caution is still needed, because the shape variation is essentially linked to changes in divertor configuration, and core confinement may be affected by local recycling. Analysis of Mark IIGB data suggests also the importance of the plasma elongation κ [7]. Therefore further experiments are under preparation to separate the three parameters δ_{up}, δ_{low} and κ. In particular, it is important to widen the lower triangularity range.

4. Impurity Seeding

A new series of radiative mantle studies [8, 9] have been undertaken by means of impurity seeding in both L and ELMy H-mode plasmas. Detailed results are presented elsewhere in this conference [10]. The most interesting new result is obtained in ELMy H-mode discharges: tailoring the influx rate of D_2 and Ar after the main gas puffing phase (the so-called "after-puff" regime) results in high density, $f_{GW} > 0.9$, and high performance, $f_{GW} \times H_{ITER89p} > 0.9$, sustained for a duration of about 3 τ_E. On a longer time scale performance is maintained, despite a rollover in density. Future work will focus on improving the stationarity of this scenario.

5. Neoclassical Tearing Modes

Neoclassical tearing modes (NTM) studies are of key importance in preparation of ITER, since they are expected to limit the achievable fusion performance in the reference ELMy H-mode regime [11]. Consequently a strong emphasis has been put on these studies during the recent experimental Campaigns and there has been much progress, as reported below.

Building on experiments performed in 1999 to study the effect of plasma shape on NTM stability, a scan in plasma triangularity has been performed. Using empirical scalings to correct to constant ρ^* and collisionality it is found that triangularity has a stabilising effect (raises the 3,2 NTM beta-threshold) while elongation appears to be destabilising. However scaling to local variables (which reflect the underlying NTM stability), or scaling to constant B_T field and n_{GW}, it is found that the dependence of the NTM threshold on triangularity becomes weaker.

The control of (3,2) NTM thresholds via ICRH and ICCD has been studied. Several Ion Cyclotron heating or CD schemes have the well-known effect of giving rise to large amplitude long period sawteeth, resulting in substantially reduced NTM thresholds. Studies conducted recently at JET show that an appropriate tuning of the ICRH deposition scheme (outside the inversion radius), or of the ICCD phasing, allows, via an appropriate control of sawteeth, the (3,2) NTM threshold to be significantly raised. These constitute promising tracks for developing active control tools in view of raising the performance of ITER plasmas. A fuller discussion of these, and other, NTM results can be found in [12].

6. Improvements in Lower Hybrid Coupling

One important recent result is the significant improvement in Lower Hybrid (LH) power coupling on JET plasmas. This was achieved by using simultaneously various techniques:
- *CD_4 puffing*: this allows a sufficiently-dense SOL plasma to develop in front of the LH grill. Up to the maximum CD_4 rate (10^{22} electron/s), the puffing seems to affect only the edge characteristics, leaving plasma core parameters unchanged;
- *plasma shape optimisation*: the poloidal asymmetry in the SOL in front of the LH grill is minimised, thus allowing for a more homogeneous reflection pattern; the distance between the last closed flux surface and the poloidal limiter can be maintained as small as 4-5 cm with the LH grill only 5-6 mm in the limiter shadow;
- *control of fast ions*: the hydrogen concentration in the discharge is controlled, to avoid sputtering of metallic impurities from the LH grill by ICRF generated fast ions.

As a result of these developments the coupling of LH was greatly improved. Good conditions are maintained in the preheat and in the main heating phase of Optimised Shear (OS) plasmas, where a total power in excess of 20 MW (mostly from NBI) is coupled to the plasma yielding H-modes (generally Type III ELMs) and ITBs. A reflection coefficient as low as R ≤ 5–6 % has allowed coupling of up to 3.8 MW of LH power during the preheat phase of OS plasmas and up to 3 MW during the high power phase both on H-mode and ITB plasmas. No case of high-Z radiation from the LH grill has been detected. The remaining limitations in LH power coupling are thought to be mainly due to technical characteristics of the system and to grill conditioning.

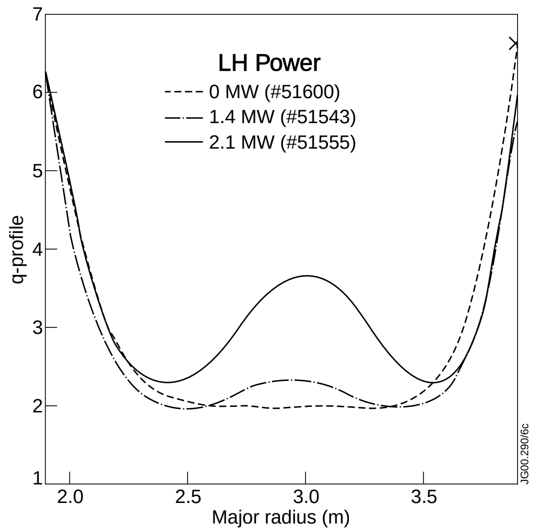

Fig. 4: Effect of LH during the pre-heat phase on the q profile shear reversal from MSE data plus equilibrium reconstruction.

7. Study of Internal Transport Barriers

Improved LH coupling turns out to play a key role in the development of ITB scenarios on JET. Indeed, a first demonstration of the benefit of LH preheat in ITB scenarios had been made at the end of 1999: high performance could be obtained at a lower threshold power [13]. The recently improved LH operation led to a systematic use of LH during the preheat phase, and to demonstrate how ITB scenarios depend on the target q profile, in close relation to the amount of LH power coupled during the plasma current ramp-up. Fig. 4 shows the strong shear reversal which can be obtained when P_{LH} increases. These studies are supported by progress in diagnostics: q profiles are routinely available, both via polarimetry and Motional Stark Effect (MSE), in combination with the equilibrium reconstruction code EFIT [14, 15].

Following this progress, the recent "advanced tokamak" activity on JET has focused mainly on the role played by the target current profile on the triggering of the ITBs. Therefore, the "pre-heat" phase has been tailored carefully in order to either slow down the plasma current inward diffusion occurring during the current ramp-up (use of additional heating power) and/or superimpose an externally driven non-inductive current fraction (use of LHCD) [13, 16, 17]. The results reported here confirm the crucial role played by the target q-profile in triggering the ITB.

A significant decrease of the power requested during the main heating phase to trigger an ITB (ITB power threshold) is obtained when either LHCD or off-axis ICRH power is used during the main heating phase. This effect also exists when faster plasma current ramp-up rates are used (in absence of any additional heating or current drive). Application of on-axis ICRH power (up to a level of 2.5 MW) does not lead to such a reduction of the ITB power threshold.

Moreover, LHCD power (at the same power level) is more "efficient" in reducing the threshold than off-axis ICRH. Fig. 5 displays some of the present JET database on ITB power threshold: discharges using 2 - 2.5 MW of LHCD during the preheat phase triggered ITBs at almost half the power required in the non-preheat discharges. This figure also shows that the linear dependence of the power threshold with the toroidal magnetic field reported previously [18] does not apply to the discharges using a pre-heat phase.

Moreover, in the operational range 2.6T < B_T < 3.5 T investigated so far, ITBs on T_e are already observed during the LHCD pre-heat phase, above P_{LHCD}~ 2MW ($n_e(0)$~1-1.5 10^{19} m^{-3}). $T_e(0)$ then reaches 10 keV. These barriers experience sawtooth-like inner collapses as time goes on.

The LHCD pre-heating has been used to achieve high performance reversed shear regimes, and significant progress was made with respect to the previously reported low shear ITBs in two directions:

1. the *fusion performance* in terms of H-factor, β_N and neutron rates was optimised. The present best discharges achieved 4.1 10^{16} neutrons/s at 3.4T, with a stored energy W_{dia} = 11 MJ at an edge-q value of q_{95} = 5. The confinement enhancement factor reached H_{89} = 3 together with a record β_N of 2.4 (the previous record was β_N ~2 at that magnetic field). This improved performance is due to the development of an extremely broad barrier (the ITB footpoint is around ρ~0.9, and T_i is above 25 keV up to mid-radius). The resulting Q_{DT}^{eq} reaches 0.5, Q_{DT}^{eq} being defined here as the ratio of the equivalent fusion power to the maximum additional power applied during the pulse (no credit is taken from dW_{dia}/dt or power ramp down). The time sequence of such discharges shows an early T_e ITB during the LHCD pre-heat phase, which turns into a global barrier (T_e, T_i and n_e) as the main ICRH+NBI power is applied. This ITB then expands extremely fast, being terminated by a giant type-I ELM and a disruption.

2. *the steady-state capability* of such regimes is being investigated. In the present campaign, the ITBs operating at a high fraction of non-inductive current drive were achieved through keeping q_{95} in the range 5-6, in order to limit the fusion performance degradation. Fig. 6 shows an attempt at 3.4 T, 1.5 MA, where a reversed shear ITB was sustained during the whole high power phase at 1.5 < β_p < 1.65. The loop voltage dropped to zero and was maintained at this value with a nearly constant l_i for 1 second. This phase ended when the ITB experienced a rollover triggered by MHD activity. The first non-inductive current analysis indicates that the bootstrap current fraction was 40%, as was the NB driven current fraction, confirming that the major part of the outer current profile (ρ > 0.5) is non-inductive. Such regimes give a promising basis for stationary discharges with ITBs in JET. The use of LHCD power during the high power phase is expected to provide the additional non-inductive current capability as well as a certain current profile control capability.

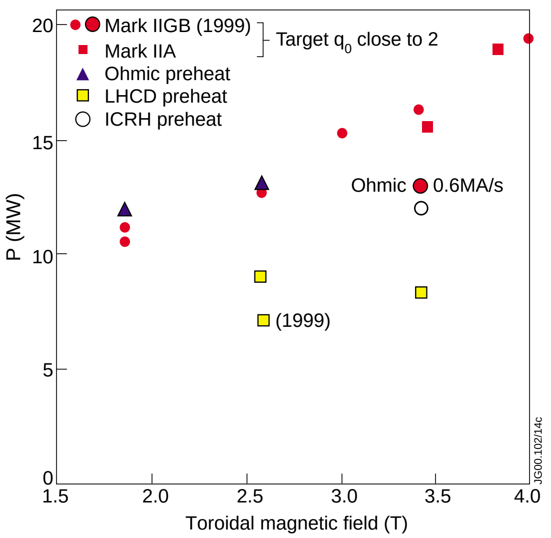

Fig. 5 : ITB power threshold versus B_T, for non preheated discharges (dI_p/dt of 0.4MA/s), as well as for discharges using LHCD or off-axis ICRH preheat. A specific 1999 discharge using a fast plasma current ramp rate (0.6MA/s) is also displayed.

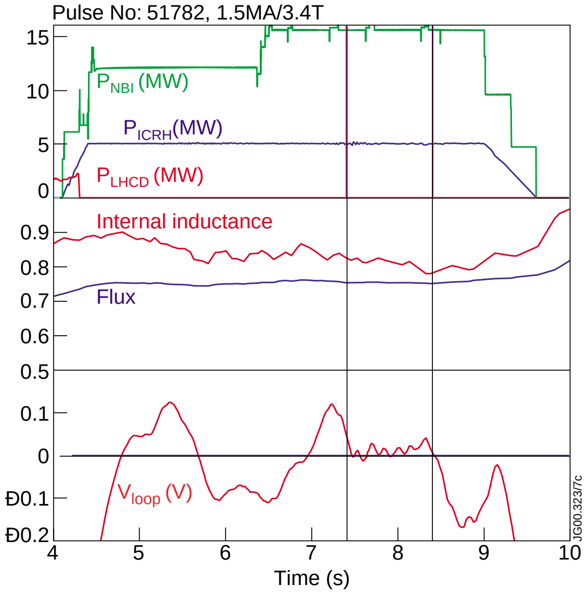

Fig. 6 : Time evolution of injected powers, internal inductance, flux and loop voltage (pulse 51782), illustrating steady-state potential.

8. Programme 2001-2002

Further experimental campaigns will be conducted during the years 2001 and 2002, with a persistent focus on ITER physics. Some enhancements will be put in operation during the 2001 and 2002 shutdowns: additional or improved diagnostics and control systems (new external field correction coils, upgraded reflectometry, new or improved edge diagnostics such as LIDAR, reciprocating probes, quartz-microbalance, etc…); upgrade of one NB injector providing ~7.5MW additional heating by end 2002 and testing of prototype devices aimed at improved RF coupling.

9. Longer Term Perspectives : the JET-EP Project

The JET-Enhanced Performance (JET-EP) project received a positive recommendation by the EURATOM-Fusion Consultative Committee (CCE-Fu) on 11 July 2000. The main objectives of the proposed enhancements are: (1) to consolidate the preparation of ITER operating scenarios by increasing the present operating domain of ELMy H-modes (the reference mode of ITER operation) and further expanding the range of operating scenarios and (2) to support design choices in key areas of ITER subsystems which could be finalised even after the start of construction.

It is planned to install a third Neutral Beam Injection box with 4 sources (7.5 MW, upgradeable to 8 sources, 15 MW), a 5 MW ECRH system (110 GHz, 1 MW gyrotrons) and a new, ITER-like, ICRH antenna, delivering 8 MW. To withstand the increased heating power and to access higher triangularity plasmas, a modified divertor will be installed. Upgrades of the diagnostic systems are also foreseen. High Resolution Thomson Scattering, additional CXRS and MSE systems, new DT diagnostics and a revised set of divertor diagnostics are presently under discussion. Most of the JET-EP elements should be commissioned by the end of 2004. The increased input power (up to 50 MW) will allow access to high β_N at high B_T for further developments of the ELMy H-mode scenarios, NTM scaling and stabilisation, and development of AT scenarios.

Acknowledgements

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ANNEX

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